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SOUTHERN COMPANY
Energy to Serve Your World™

January 8, 2003

Docket Nos. 50-425

Log: NL-03-0079
LCV-1643

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

VOGTLE ELECTRIC GENERATING PLANT
LICENSEE EVENT REPORT 2-2002-002
STEAM GENERATOR LEVEL CONTROL
PROBLEMS LEAD TO MANUAL REACTOR TRIP

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73, Southern Nuclear Operating Company hereby submits a Vogtle Electric Generating Plant licensee event report for a condition that occurred on November 13, 2002.

Please contact this office if you have any questions.

Sincerely,

Jeffrey T. Gasser

Jeffrey T. Gasser

JTG/NJS

Attachment: LER 2-2002-002

cc: Southern Nuclear Operating Company
Mr. G. R. Frederick
Mr. M. Sheibani
Document Services – RTYPE CVC7000

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. F. Rinaldi, Vogtle Project Manager, NRR
Mr. J. Zeiler, Senior Resident Inspector, Vogtle

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7/31/2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503 If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to,

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4 TITLE
STEAM GENERATOR LEVEL CONTROL PROBLEMS LEAD TO MANUAL REACTOR TRIP

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S) 05000
11	13	2002	2002	002	00				FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE	1
10. POWER LEVEL	21

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § : (Check all that apply)

	20 2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50 73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20 2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73 71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73 71(a)(5)
	20-2203(a)(2)(ii)		50 36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20-2203(a)(2)(iii)		50 46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20 2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Mehdi Sheibani, Nuclear Safety and Compliance	(706) 826-3209

13 COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i e , approximately 15 single-spaced typewritten lines)

On November 13, 2002, power ascension was in progress following a refueling outage. At approximately 0354 EST, the licensed operator designated as the steam generator water level control operator (SGWLCO) began to transfer steam generator feedwater control from the bypass feedwater regulating valves (BFRVs) to the main feedwater system regulating valves (MFRVs). A series of water level transients commenced in each of the four steam generators (SGs) that culminated in the SG #3 water level increasing to its Hi-Hi level setpoint at 0405 EST. The main feedwater system isolated, the main feedwater pump tripped, and the auxiliary feedwater system actuated, as designed. As SG water levels decreased to their low level setpoints, a manual reactor trip was initiated at 0405 EST, and SG water levels were stabilized in Mode 3 (hot standby).

The root cause of this event was the failure of the SGWLCO to follow procedure by having all BFRVs and MFRVs open simultaneously. A secondary cause of this event was less than adequate supervisory oversight by the Unit Shift Supervisor in verifying procedure compliance during feedwater control operations. The policy for implementing unit operating procedures is being reviewed to determine if the level of flexibility and/or interpretation is appropriate. Expectations of command and control issues are also being evaluated.

A. REQUIREMENT FOR REPORT

This event is reportable per 10 CFR 50.73 (a)(2)(iv) because unplanned engineered safety feature and unplanned reactor protection system actuations occurred.

B. UNIT STATUS AT TIME OF EVENT

At the time of this event, Unit 2 was in power ascension in Mode 1 (power operations) at 21 percent of rated thermal power. The main feedwater system was in service to all four steam generators feeding through the bypass feedwater regulating valves (BFRVs). Personnel were making preparations to synchronize the generator to the grid. There was no inoperable equipment that contributed to the occurrence of this event.

C. DESCRIPTION OF EVENT

On November 13, 2002, power ascension was in progress following a refueling outage. At approximately 0354 EST, the licensed operator designated as the steam generator water level control operator (SGWLCO) began to transfer steam generator feedwater control from the BFRVs to the main feedwater system regulating valves (MFRVs). The SGWLCO incorrectly implemented the procedure for this transfer of control, initiating a series of water level transients in each of the four steam generators (SGs). In response, the SGWLCO closed and re-opened the MFRVs and decreased and increased the main feedwater pump speed. At 0403 EST, the last of the BFRVs was closed. At 0404 EST, the SG #3 water level was rising rapidly and MFRV #3 was closed. However, at 0405 EST, the SG #3 water level increased to greater than the 86% Hi-Hi level setpoint, initiating a main feedwater system isolation, a main feedwater pump trip, and an auxiliary feedwater system (AFW) actuation, as designed. As SG water levels decreased to their low level setpoints, a manual reactor trip was initiated at 0405 EST, and SG water levels were stabilized in Mode 3 (hot standby).

D. CAUSE OF EVENT

The root cause of this event was the failure of the SGWLCO to follow procedure 12004-C, "Power Operation (Mode 1)." Procedure step 4.1.26 requires that feedwater control from BFRVs to MFRVs be transferred one loop at a time. The SGWLCO interpreted this to allow all BFRVs and MFRVs to be opened simultaneously and under manual control prior to closing the four BFRVs one at a time. Also contributing to the difficulty in controlling water levels was the

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SGWLCO's use of the main feedwater pump speed to control water input. At a given power level, pump speed should remain constant. These factors resulted in the inability to adequately control SG water levels.

A secondary cause of this event was less than adequate supervisory oversight by the Unit Shift Supervisor (USS) in verifying procedure compliance during feedwater control operations.

There were no characteristics of the work location that contributed to the occurrence of these errors by the control room personnel involved.

E. ANALYSIS OF EVENT

The main feedwater system isolated, the operating main feedwater pump tripped, and the auxiliary feedwater system actuated as designed following the receipt of the SG high water level signal. With the main feedwater system isolated, the main feedwater pump tripped, and reactor power at 21%, control room personnel acted appropriately to manually trip the reactor and prevent a challenge to the automatic trip actuation circuitry. Based on these considerations, there was no adverse effect on plant safety or on the health and safety of the public as a result of this event.

This event does not represent a safety system functional failure.

F. CORRECTIVE ACTIONS

1) This event was reviewed with the operating crew assigned to re-start the plant and the unit was successfully returned to power.

2) This event will be addressed in licensed operator continuing training by March 1, 2003, emphasizing the effects of over-controlling feed flows via changes in pump speed, and the need to monitor steamflow/feedflow mismatch in this situation. Expectations of command and control issues will also be addressed.

3) By March 1, 2003, the Operations department will review its policy for implementing unit operating procedures to determine if the level of flexibility and/or interpretation is appropriate. Additionally, operating experience of similar events will be reviewed to determine if enhancements should be made to operating procedures.

4) The SGWLCO and the USS are no longer employed with Southern Nuclear Operating Company.

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G. ADDITIONAL INFORMATION

1) Failed Components:
None

2) Previous Similar Events:
LER 50-424/2002-003 dated June 13, 2002. The corrective actions for this June 13, 2002, LER were specific for preventing a recurrence of the April 20, 2002, reactor trip. These corrective actions were not general enough to prevent the reactor trip of November 13, 2002.

3) Energy Industry Identification System Code:
Main Feedwater System – SJ
Auxiliary Feedwater System – BA

STP Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

January 13, 2003
NOC-AE-03001449
10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Unit 1
Docket No. STN 50-498
Unit 1 Cycle 11 End of Life Moderator Temperature Coefficient Limit Report

Reference: Letter, J. J. Sheppard to U.S. Nuclear Regulatory Commission, "End of Life Moderator Temperature Coefficient," dated October 31, 2002 (NOC-AE-02001425)

As a condition for approval of the conditional elimination of the most negative end of life moderator temperature coefficient measurement technical specification change as stated in the referenced correspondence, STP committed to submit the following information for the first three uses of this methodology at STP:

1. A summary of the plant data used to confirm that the Benchmark Criteria of Table 3-2 of WCAP-13749-P-A, *Safety Evaluation Supporting the Conditional Elimination of the Most Negative EOL Moderator Temperature Coefficient Measurement,* have been met; and,

2. The Most Negative EOL Moderator Temperature Coefficient Limit Report (as found in Appendix D of WCAP-13749-P-A).

The information is attached. If there are any questions regarding this information, please contact Mr. Duane Gore at (361) 972-8909.

D. C. Leazar

D.A. Leazar
Manager,
Nuclear Fuel and Analysis

Attachments:
1. Plant Data Used to Confirm Benchmark Requirements
2. Most Negative End of Life Moderator Temperature Coefficient Limit Report for South Texas Unit 1, Cycle 11

cc:

(paper copy)

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Attachment 1

Plant Data
Used to Confirm Benchmark Requirements

Plant Data Used to Confirm Benchmark Requirements are Satisfied

This attachment presents a comparison of the South Texas Unit 1 Cycle 11 core characteristics with the requirements for use of the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement Methodology and presents plant data that support that the Benchmark Criteria presented in WCAP-13749-P-A are met.

The Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement Methodology is described in WCAP-13749-P-A. This report was approved by the NRC with two requirements:

- only PHOENIX/ANC calculation methods are used for the individual plant analyses relevant to determinations for the EOL MTC plant methodology, and
- the predictive correction is reexamined if changes in core fuel designs or continued MTC calculation/measurement data show significant effect on the predictive correction.

The PHOENIX/ANC calculation methods were used for the South Texas Unit 1, Cycle 11, core design and relevant analyses. Also, the Unit 1, Cycle 11, core design does not represent a major change in core fuel design. Therefore, the Predictive Correction of –3 pcm/°F remains valid for this cycle. The Unit 1, Cycle 11, core meets both of the above requirements.

A description of the data collection and calculations required to complete the Table 3 Worksheet of the Most Negative Moderator Temperature Coefficient Limit Report is presented. Then the following data tables are provided:

- Table 1 - Benchmark Criteria for Application of the 300 ppm MTC Conditional Exemption Methodology (per WCAP-13749-P-A)
- Table 2 - Flux Map Data: Assembly Powers and Core Tilt Criteria
- Table 3 - Core Reactivity Balance Data
- Table 4 - Low Power Physics Test Data (Beginning of Cycle, Hot Zero Power): Isothermal Temperature Coefficient (ITC)
- Table 5 - Low Power Physics Test Data (Beginning of Cycle, Hot Zero Power): Individual Control Bank Worth

Table 1
Benchmark Criteria for Application of the 300 ppm MTC Conditional Exemption Methodology (per WCAP-13749-P-A)

Parameter	Criteria
Assembly Power (Measured Normal Reaction Rate)	± 0.1 or 10 %
Measured Incore Quadrant Power Tilt (Low Power)	± 4 %
Measured Incore Quadrant Power Tilt (Full Power)	± 2 %
Core Reactivity (Cb) Difference	± 1000 pcm
BOL HZP ITC	± 2 pcm/°F
Individual Control Bank Worth	± 15 % or ± 100 pcm
Total Control Bank Worth	± 10 %

Table 2
Flux Map Data:Assembly Powers and Core Tilt Criteria

Flux Map Number	Assembly Power: Measured to Predicted Error		Benchmark Criteria: Requirement	Benchmark Criteria: Criteria Satisfied	Measured Incore Quadrant Power Tilt: Power Tilt		Benchmark Criteria: Requirement	Benchmark Criteria: Criteria Satisfied
111001	% Diff	4.1	% Diff within ± 10% OR M-P within ± 0.1	Yes	Max	1.0132	Maps at < 90% Reactor Power Max Power Tilt ≤ 1.04 And Min Power Tilt ≥ 0.96 OR Maps at > 90% Reactor Power Max Power Tilt ≤ 1.02 And Min Power Tilt ≥ 0.98	Yes
	Meas – Pred	0.049			Min	0.98164		
111002	% Diff	4.5		Yes	Max	1.00361		Yes
	Meas – Pred	0.048			Min	0.99612		
111003	% Diff	4.4		Yes	Max	1.00516		Yes
	Meas - Pred	0.051			Min	0.99206		
111004	% Diff	-3.9		Yes	Max	1.00509		Yes
	Meas - Pred	-0.047			Min	0.99206		
111005	% Diff	-3.7		Yes	Max	1.00385		Yes
	Meas - Pred	-0.045			Min	0.99293		
111006	% Diff	-3.4		Yes	Max	1.00403		Yes
	Meas - Pred	-0.043			Min	0.99458		
111007	% Diff	9.8		Yes	Max	1.00151		Yes
	Meas - Pred	-0.041			Min	0.99857		
111008	% Diff	9.6		Yes	Max	1.00122		Yes
	Meas - Pred	0.04			Min	0.99871		
111009	% Diff	10.1		Yes	Max	1.00173		Yes
	Meas - Pred	0.043			Min	0.99848		
111010	% Diff	10 3		Yes	Max	1.00787		Yes
	Meas - Pred	0.049			Min	0.99456		
111011	% Diff	10.2		Yes	Max	1.00258		Yes
	Meas - Pred	0.045			Min	0.99744		
111012	% Diff	11.4		Yes	Max	1.00191		Yes
	Meas - Pred	0.052			Min	0.99899		
111013	% Diff	11.6		Yes	Max	1.00049		Yes
	Meas - Pred	0.053			Min	0.99972		
111014	% Diff	7.1		Yes	Max	1.00352		Yes
	Meas - Pred	0.038			Min	0.99605		
111015	% Diff	7.7		Yes	Max	1.00287		Yes
	Meas - Pred	0.035			Min	0.99874		
111016	% Diff	7.1		Yes	Max	1.00639		Yes
	Meas - Pred	0.04			Min	0.99179		
111017	% Diff	7.6		Yes	Max	1.00767		Yes
	Meas - Pred	0.042			Min	0.98997		
111018	% Diff	7.5		Yes	Max	1.00704		Yes
	Meas - Pred	0.044			Min	0.98888		

Table 3
Core Reactivity Balance Data

Surveillance Date/Time		Core Reactivity Difference (Critical boron)		
		Reactivity Deviation (pcm)	Benchmark Criteria	
			Requirement	Satisfied
10/30/01	16:58	69.3	Reactivity Deviation within ± 1000 pcm	Yes
11/27/01	14:51	-75.6		Yes
12/18/01	15:39	-235.0		Yes
01/15/02	16:30	-275.2		Yes
02/13/02	14:35	-328.3		Yes
03/11/02	16:06	-335.4		Yes
04/10/02	16:03	-385.4		Yes
05/08/02	11:27	-408.7		Yes
06/03/02	15:47	-370.6		Yes
07/02/02	15:00	-331.5		Yes
07/30/02	16:13	-281.3		Yes
08/27/02	15:01	-265.3		Yes
09/24/02	16:06	-202.8		Yes
10/22/02	15:10	-172.0		Yes
11/27/02	15:23	-35.7		Yes
12/17/02	14:17	-1.4		Yes

Table 4
Low Power Physics Test Data
(Beginning of Cycle, Hot Zero Power):
Isothermal Temperature Coefficient (ITC)

	Measured (pcm/°F)*	Predicted (pcm/°F)*	Error (Measured – Predicted) (pcm/°F)*	Benchmark Criteria	
				Requirement	Satisfied
BOC HZP ITC	-1.66	-2.35	0.69	ITC Error within ±2 pcm/°F	Yes

*Note: 1 pcm = 1 x 10^{-5} ΔK/K

Table 5
Low Power Physics Test Data
(Beginning of Cycle, Hot Zero Power):
Individual Control Bank Worth

Bank	Measured (pcm)*	Predicted (pcm)*	Δ Error (pcm)*	% Error	Benchmark Criteria	
					Requirement	Satisfied
Shutdown Bank A	278.6	272.1	6.5	2.4%	% Error within ±15% OR Δ Error within ±100 pcm	Yes
Shutdown Bank B	799.6	775.3	24.3	3.1%		Yes
Shutdown Bank C	413.4	397.3	16.1	4.1%		Yes
Shutdown Bank D	398.7	389.6	9.1	2.3%		Yes
Shutdown Bank E	487.0	483.1	3.9	0.8%		Yes
Control Bank A	791.6	776.4	15.2	2.0%		Yes
Control Bank B	687.2	656.1	31.1	4.7%		Yes
Control Bank C	862.7	845.4	17.3	2.1%		Yes
Control Bank D	540.1	516.4	23.7	4.6%		Yes
Total Control Bank Worth	5258.9	5111.7	147.2	2.9%	% Error within ±10%	Yes

*Note: 1 pcm = 1 x 10^{-5} ΔK/K

Attachment 2

Most Negative End of Life Moderator Temperature Coefficient Limit Report for South Texas Unit 1, Cycle 11

Most Negative End of Life Moderator Temperature Coefficient Limit Report for South Texas Unit 1, Cycle 11

(Measured 300 ppm Burnup, as per WCAP-13749-P-A, Appendix D)

PURPOSE:

The purpose of this document is to present cycle-specific best estimate data for use in confirming the most negative end of life moderator temperature coefficient (MTC) limit in Technical Specification 3.1.1.3. This document also summarizes the methodology used for determining if a HFP 300 ppm MTC measurement is required.

PRECAUTIONS AND LIMITATIONS:

The EOL MTC elimination data presented in this document apply to South Texas Unit 1 Cycle 11 only and may not be used for other operating cycles.

The following reference is applicable to this document:

Fetterman, R. J., Slagle, W. H., *Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement*, WCAP-13749-P-A, March, 1997.

PROCEDURE:

All core performance benchmark criteria listed in Table 1 must be met for the current operating cycle. These criteria are confirmed from startup physics test results and routine HFP boron concentration and flux map surveillance performed during the cycle.

If all core performance benchmark criteria are met, then the Revised Predicted MTC may be calculated per the algorithm given in Table 2. The required cycle specific data are provided in Table 2 and Figure 1. This methodology is also described in Reference 1. If all core performance benchmark criteria are met, and the Revised Predicted MTC is less negative than COLR Limit 2.3.3, then a measurement is not required.

Note that Figure 1 is not entirely linear. However, the deviation is slight enough that linear interpolation between adjacent points from the data at the bottom of the Figure is acceptable.

Table 1
Benchmark Criteria for Application of the 300 ppm MTC
Conditional Exemption Methodology

Parameter	Criteria
Assembly Power (Measured Normal Reaction Rate)	± 0.1 or 10 %
Measured Incore Quadrant Power Tilt (Low Power)	± 4 %
Measured Incore Quadrant Power Tilt (Full Power)	± 2 %
Core Reactivity (Cb) Difference	± 1000 pcm
BOL HZP ITC	± 2 pcm/°F
Individual Control Bank Worth	± 15 % or ± 100 pcm
Total Control Bank Worth	± 10 %

Table 2

Algorithm for Determining the Revised Predicted Near-EOL 300 ppm MTC

The Revised Predicted MTC = Predicted MTC + AFD Correction – 3 pcm/°F
where:

Predicted MTC is calculated from Figure 1 at the burnup corresponding to the measurement of 300 ppm at RTP conditions,

AFD Correction is the more negative value of:

$$\{\ 0 \text{ pcm/°F}, (\ \Delta AFD * AFD\ Sensitivity\ )\ \}$$

ΔAFD is the measured AFD minus the predicted AFD from an incore flux map taken at or near the burnup corresponding to 300 ppm.

AFD Sensitivity = 0.05 pcm / °F / ΔAFD

Predictive Correction is –3 pcm/°F, as included in the equation for the Revised Predicted MTC.

Table 3
Worksheet for Calculating the Predicted Near-EOL 300 ppm MTC

Unit: 1, Cycle 11 **Date:** 12/17/2002 **Time:** 1525

Reference for Cycle-Specific MTC Data:

Letter from T.D. Croyle, Westinghouse, to D.F. Hoppes, STPNOC, *[STPEGS] Unit 1 Cycle 11 Most Negative Moderator Temperature Cofficient Limit Report*, dated 19 Nov 2002, ST-UB-NOC-02002311.

Part A. Predicted MTC

A.1	Cycle Average Burnup Corresponding to the HFP ARO equilibrium xenon C_B of 300 ppm.	15171.8	MWD/MTU
A.2	Predicted HFP ARO MTC corresponding to burnup (A.1)	-34.96	pcm/°F

Part B. AFD Correction

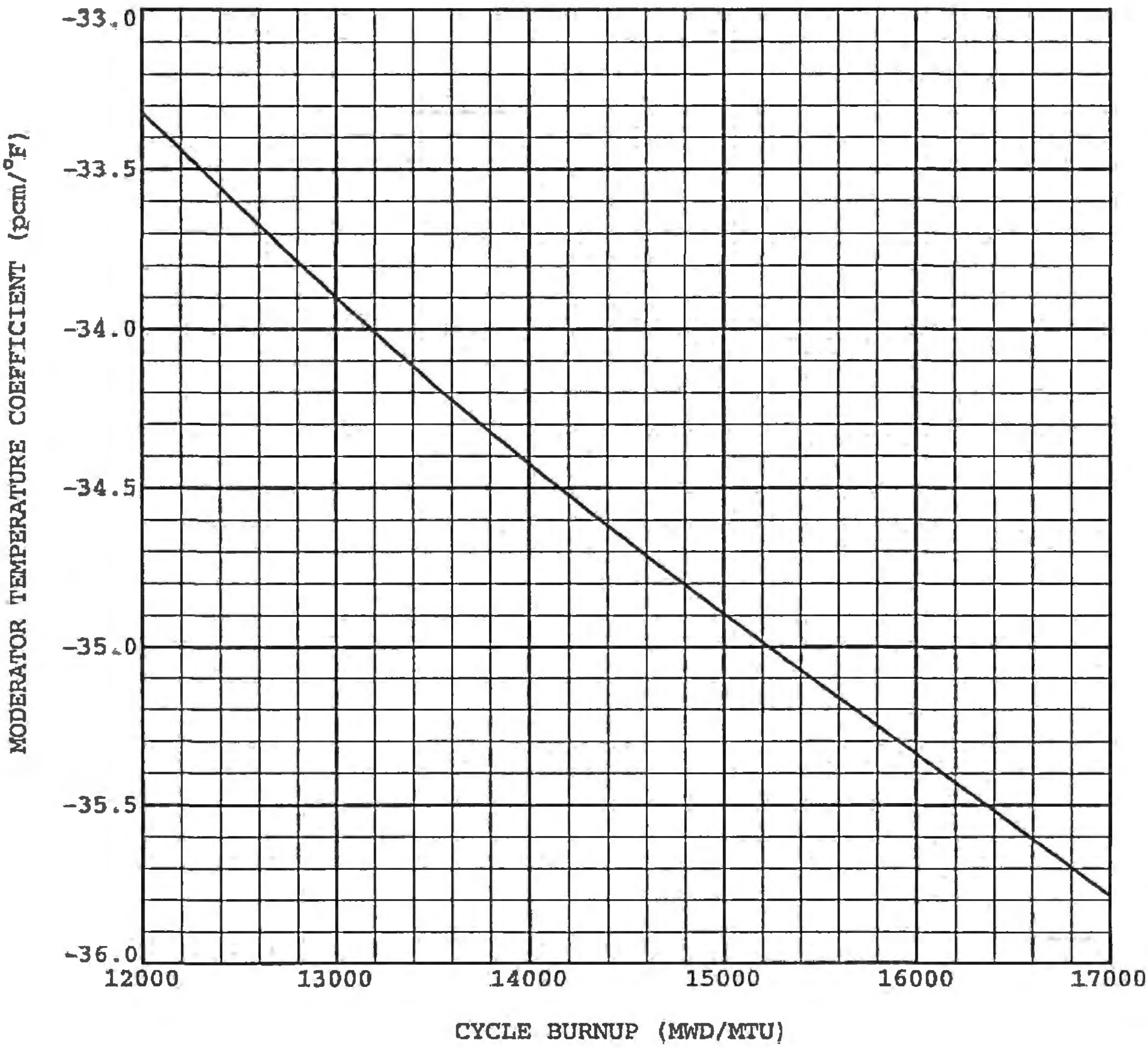
B.1	Burnup of most recent HFP, equilibrium conditions incore flux map	15200.9	MWD/MTU
B.2	Measured HFP AFD at burnup (B.1) Reference incore flux map: ID: 111018 Date: 12/17/02	-2.02	% AFD
B.3	Predicted HFP AFD at burnup (B.1)	-3.07	% AFD
B.4	MTC Sensitivity to AFD	0.05	pcm/°F/ΔAFD
B.5	AFD Correction, more negative of { 0 pcm/°F, B.4 *(B.2 – B.3)}	0	pcm/°F

Part C. Revised Prediction

C.1	Revised Prediction (A.2 + B.5 – 3)	-37.96	pcm/°F
C.2	Surveillance Limit (COLR 2.3.3)	-53.6	pcm/°F

If C.1 is less negative than C.2, then the HFP 300 ppm MTC measurement is not required per Specification 4.1.1.3.

Figure 1
Predicted HFP FOP 300 ppm MTC vs. Cycle 11 Burnup



Cycle Burnup (MWD/MTU)	Moderator Temperature Coefficients (pcm/°F)
12000	-33.32
14000	-34.43
16000	-35.34
17000	-35.79

Table 4
Data Collection and Calculations Required to Complete the Table 3 Worksheet
of the Most Negative Moderator Temperature Coefficient Limit Report

Data at the 300 ppm Boron Point

- RCS Boron at 300 ppm at 14:24 on 12/16/02.
- Burnup at 300 ppm: 15171.8 MWD/MTU **(A.1)**
- Predicted MTC: -34.96 pcm/°F **(A.2)**

Data from Last Flux Map:

- Flux Map Number: 111018 **(B.2)**
- Reactor Power 100% RTP
  ***Note:** The monthly flux map was performed at about the same time the unit reached the 300 ppm concentration value. Data from this flux map was used for the AFD Correction.*
- Burnup 15200.9 MWD/MTU **(B.1)**
- Measured Axial Offset (MAO): -2.02% **(B.2)**
  ***Note:** The Westinghouse BEACON computer code (similar to the Westinghouse INCORE code) determines Axial Offset (AO), not Axial Flux Difference (AFD). Therefore, the AO must be converted to AFD before use. The relationship between AO and AFD is*

  AFD = Axial Offset * Fractional Power

- Axial Flux Difference
  Lower Predicted AO (LPAO): -2.91% at 14000 MWD/MTU
  Higher Predicted AO (HPAO): -3.17% at 16000 MWD/MTU
  Predicted AO (PAO) =

$$PAO = \frac{B/U_{@\,Measued\,AO} - B/U_{@\,Lower\,Predicted\,AO}}{B/U_{@\,Higher\,Predicted\,AO} - B/U_{@\,Lower\,Predicted\,AO}} \times (HPAO - LPAO) + LPAO$$

PAO = (15200.9 – 14000)/(16000 – 14000) * (-3.17% + 2.91%) - 2.91% = -3.07% **(B.3)**

Δ AFD = (MAO-PAO) * 100%
= (-2.02% + 3.07%) * 100%
= 1.05%

Table 4 (cont.)
Data Collection and Calculations Required to Complete the Table 3 Worksheet of the Most Negative Moderator Temperature Coefficient Limit Report

Determination of the Revised Predicted Moderator Temperature Coefficient (MTC)

AFD Sensitivity: 0.05 pcm/°F/ ΔAFD

AFD Correction: 0 pcm/°F **(B.5)**

where: AFD Correction is the more negative of the following:

0 pcm/°F or (ΔAFD * AFD Sensitivity)
0 pcm/°F or (1.05% * 0.05 pcm/°F/ ΔAFD)
0 pcm/°F or 0.053 pcm/°F
∴0 pcm/°F

Revised Predicted MTC = Predicted MTC + AFD Correction - 3 pcm/°F
= -34.96 pcm/°F + 0.0 pcm/°F – 3 pcm/°F
= -37.96 pcm/°F **(C.1)**